VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 15, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 01-634
NL&OS/ETS R0
Docket No. 50-280
License No. DPR-32

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT 1
PROPOSED RISK-INFORMED TECHNICAL SPECIFICATIONS CHANGE
FIVE YEAR EXTENSION OF TYPE A TEST INTERVAL

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests an amendment to Facility Operating License Number DPR-32 in the form of a change to the Technical Specifications for Surry Power Station Unit 1. The proposed change will permit a one-time five-year extension of the ten-year performance based Type A test interval established in NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995.

This Technical Specification change has been prepared in accordance with the guidance provided in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk Informed Decisions on Plant Specific Changes to the Licensing Basis." A discussion of the proposed change and the associated supporting risk assessment are included in Attachments 1 and 2, respectively. A marked-up Technical Specifications page that reflects the proposed change and a revised Technical Specifications page that incorporates the proposed change are provided in Attachments 2 and 3, respectively.

The proposed changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards. In addition, the proposed change has been determined to qualify for categorical exclusion from an environmental assessment as

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set forth in 10 CFR 51.22(c)(9). The basis for these determinations is included in Attachment 1.

To permit effective outage planning, it is requested that the NRC approve the proposed Technical Specification changes by August 1, 2002. Once approved the amendment will be implemented within 30 days. Should you have any questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Commitments made is this letter: None

Attachments:
1. Discussion of Change
2. Risk Assessment
3. Mark-up of Technical Specifications
4. Proposed Technical Specifications

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SN: 01-634
Docket No.: 50-280
Subject: Proposed R-I TS Change
Five Year Ext. of Type A Test Interval

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 15th day of October, 2001.

My Commission Expires: 3-31-04.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Change

Surry Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

Discussion of Change

Introduction

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests a change to the Surveillance Requirements in Section 4.4 of the Technical Specifications for the containment structure. The proposed change will permit a one-time five-year exception for Surry Unit 1 from the requirement of NEI 94-01 (Reference 1) which specifies performance of an integrated leak rate test (ILRT) at a frequency of up to ten years with allowance for a fifteen-month extension.

The proposed change has been reviewed and it has been determined that the change qualifies for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change.

Background

Surry Power Station Unit 1's current ten year Type A test interval ends on April 23, 2002. In order to meet the interval requirements of NEI 94-01, this test must be performed during either the fall 2001 refueling outage, or using the fifteen-month extension provision, during the Spring 2003 refueling outage. By granting the proposed one-time exception, Surry would benefit by not having to perform the Type A test for a minimum of four years. Cost savings are estimated at $325,000 for elimination of the actual performance of the test. In addition, up to thirty-eight hours of critical path outage time can be eliminated by not performing the Type A test. Replacement power cost for the critical path time is estimated at $570,000.

Dominion is aware of an ongoing industry/NRC initiative to modify the existing performance-based leakage testing guidance to extend the maximum Type A test interval. Therefore, the requested exception is limited to only five years for Surry Unit 1, which is considered an adequate amount of time to complete the guidance change initiative.

Description of Change

This application for amendment to the Surry Technical Specifications proposes to revise the Technical Specification Surveillance Requirement in Section 4.4.B.1, Containment Leakage Rate Requirements. This revision takes a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by NEI 94-01. The one-time exception for Surry Unit 1 applies to the requirement of NEI 94-01 to perform an integrated leak rate test (ILRT) at a frequency of up to ten years, with allowance for a fifteen-month extension. The exception is to allow ILRT testing within fifteen years from the last ILRT, performed on April 23, 1992.

This application represents a cost beneficial licensing change. The integrated leak rate test imposes significant expense on the station while the safety benefit of performing it within ten years, versus fifteen years, is minimal as described in the attached risk assessment. The specific change to TS Section 4.4.B.1 (page 4.4.1) is as follows:

- Add the following phrase after the end of the first sentence in 4.4.B.1

  "as modified by the following exception:"

- And include the following exception to NEI 94-01

  "NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after the April 23,1992 Type A test shall be performed no later than April 22, 2007."

- Number the last sentence 4.4.B.1 as 2 and renumber 4.4.B.2 as 4.4.B.3

Safety Implications of the Proposed Change

Implementing 10 CFR 50, Appendix J, Option B:

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the Technical Specifications. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

10 CFR 50, Appendix J, was revised, effective October 26,1995, to allow licensees to choose containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements." Amendment 208 (Reference 2) was issued to Surry Power Station to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 208 modified Technical Specification Section 4.4.B which requires Type A, B and C testing in accordance with Regulatory Guide (RG) 1.163 (Reference 3). Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8–1994 (Reference 4), subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states "The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Therefore, this application does not require an exemption to Option B.

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to Type A test frequency did not result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493 (Reference 5). NUREG-1493 made the following observations with regard to decreasing the test frequency:

- “Reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing had minimal impact on public risk.”

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated leakage rate was less than $1.0L_a$) and consideration of the performance factors in NEI 94-01, Section 11.3. Based on the June 1988 and April 1992 ILRTs, the current interval for Surry Unit 1 is once every ten years.

Plant Specific Risk Assessment for the Extended ILRT Test Interval:

The risk assessment was performed in accordance with the guidelines set forth in NEI 94-01, the methodology used in EPRI TR-104285 and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, RG 1.174. In addition, the results and findings from the Surry Individual Plant Examination (IPE) and revised model are used for this risk assessment calculation.

Method of Analysis:

A simplified bounding analysis approach for evaluating the change in risk associated with increasing the interval from ten years to fifteen years for Type A test was used. Type A test measures the containment air mass and calculates the leakage from the change in mass over time. This approach is similar to that presented in EPRI TR-104285 and NUREG-1493. Namely, the analysis performed examined SPS IPE plant specific accident sequences in which the containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).

- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. [For example, a liner breach or steam generator manway leakage (EPRI TR-104285 Class 3 sequences).] Type B tests measure component leakage across pressure retaining boundaries (e.g. gaskets, expansion bellows and air locks). Type C tests measure component leakage rates across containment isolation valves.

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left opened following a plant post-maintenance test. [For example, a valve failing to close following a valve stroke test (EPRI TR-104285 Class 6 sequences).]

- Accident sequences involving containment failure induced by severe accident phenomena (EPRI TR-104285 Class 7 sequences), containment bypassed (EPRI TR-104285 Class 8 sequences) and large containment isolation failures (EPRI TR-104285 Class 2 sequences). Small containment isolation 'failure-to-seal' events (EPRI TR-104285 Class 4 and 5 sequences) were not accounted for in this evaluation. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test interval.

Conclusions:

Based on the above sequences considered, the following conclusions are made regarding the plant risk associated with extending the Type A ILRT test frequency from ten years to fifteen years:

- RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 4.37 x 10^{-8}/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from ten to fifteen years is therefore, considered non-risk significant. The calculation is included as an attachment to this Technical Specification change request.

- The one-time change to the Type A test interval from ten years to fifteen increases the risk of those associated specific accident sequences by 0.001%. In addition, the risk impact on the total integrated (fifteen year increase) plant risk above baseline, for those accident sequences influenced by Type A testing is only 0.004%. Therefore, the risk impact when compared to other severe accident risks is negligible.

10 CFR 50 *Appendix J, Option B Integrated Leak Test Information:*

A Type A test can detect containment leakage due to a loss of structural capability. All other sources of containment leakage detected in Type A test analyses can be detected by the Type B and C tests.

Previous Type A tests confirmed that the Surry Unit 1 reactor containment structure has extremely low leakage and represents an insignificant potential risk contributor to increased containment leakage. The increased leakage is minimized by continued Type B and Type C testing for penetrations with direct communication with containment atmosphere. Also, the In-Service Inspection (ISI) program and maintenance rule program require periodic inspection of the interior and exterior of the containment structure to identify degradation.

The results for the last two Type A test are reported in the following table for Surry Unit 1:

Date	As Found Leakage(*)	Acceptance Limit(**)	Test Pressure (psia)
April 23, 1992	0.396	1.0 L_a	44.46
June 26, 1988	0.5055	1.0 L_a	44.46

* This is the leakage attributable to containment leakage as well as a number of Type B and Type C leakage components being tested as part of the Type A test. The leakage is the percent (%) of containment air by weight per day.

** The total allowable "as-left" leakage is expressed in percent (%) of containment air by weight per day and is 0.75 L_a, (L_a = 0.1% of primary containment air by weight per day and is the leakage assumed in dose consequences) with 0.6 L_a, the maximum leakage from Type B and C components.

Plant Operational Performance:

During power operation, Surry Unit 1 is maintained at a subatmospheric condition. Instrumentation provides constant indication of containment pressure. If pressure rises, an alarm annunciates conditions approaching the limits allowed by the Technical Specifications. In addition, if a containment vacuum pump is required to maintain subatmospheric conditions, discharge flow rates can be monitored locally which provides additional indication of containment condition. This monitoring of the containment pressure equates to continuous on-line monitoring of the containment leakage during operation. Although not as significant as the differential pressure resulting from a Design Basis Accident, the fact that the containment can be maintained subatmospheric provides a degree of assurance of containment structural integrity (i.e., no large leak paths in the containment structure). This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination.

IWE/IWL Inservice Inspection (ISI) Program and Activities to Support ILRT:

The current regulatory requirement mandated by 10 CFR 50.55a requires licensees to implement a containment inspection program in accordance with the rules and requirements of the 1992 Edition through the 1992 Addenda of ASME Section XI, Subsections IWE and IWL, as amended in the regulation. Dominion implemented the Containment ISI Program in accordance with these rules at each of its four nuclear units. The regulatory requirement allows five years for the implementation of the first period inspections. In consideration of these rules, the Initial Period (First Period) for the performance of Containment ISI began on September 9, 1996 and ended on September 8, 2001. The subsequent periods (IWE) will comply with the normal period requirements of four years for the second period and three years for the third period of inspection program B of ASME Section XI. The subsequent IWL intervals are repeated every five years. The proposed frequency extension of ILRT requirements would have no affect upon these requirements.

The regulatory requirement additionally required that the general visual examination, IWE Category E-A, be conducted each inspection period in addition to the Code requirement of just prior to the Type A test during the interval. This general visual is similar to the visual requirement of Appendix J. The general visual examination requirement conducted each period would be maintained even if the extended ILRT frequency negates a Type A test within the normal Code ten-year interval. As such, no Code requirement (IWE, Category E-A) would be affected by the ILRT frequency extension.

The following relief requests were reviewed to assess the effect, if any, resulting from the proposed ILRT frequency extension:

- Relief Request RR-IWE2 obtained relief from Section XI of the ASME Code, 1992 Edition, 1992 Addenda, Code Items E5.10 and E5.20 which require a visual examination of metal containment seals and gaskets. The relief permits continued acceptance of containment seals and gaskets through the performance of 10 CFR 50 Appendix J testing rather than by individual visual inspection. NRC letter dated April 14, 1999 granted this relief to Surry Units 1 & 2. The proposed ILRT frequency extension only affects Type A testing. The Type B testing program remains unaffected and, therefore, the relief request remains valid and unaffected by the proposed change.

- Relief Request RR-IWE5 obtained relief from Section XI of the ASME Code, 1992 Edition, 1992 Addenda, Code Item E8.20, which requires a bolt torque or tension test for bolted connections that have not been disassembled and reassembled during the inspection interval. The relief permits leak tightness of bolted connections to be verified through the performance of 10 CFR 50 Appendix J testing. NRC letter dated April 14, 1999 granted the relief request for Surry Units 1 & 2. The proposed ILRT frequency extension only affects Type A testing only. The Type B testing program remains unaffected. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request RR-IWE8 obtained relief from Section XI of the ASME Code 1992 Edition, 1992 Addenda, Table IWE-2500-1, Category E-P, which contains examination requirements in conjunction with post repair, replacement and 10 CFR 50 Appendix J requirements. NRC letter dated March 20, 2000 granted the relief request for Surry Units 1 and 2. The relief request is administrative in nature, removing redundant Code requirements addressed by Appendix J and eliminating unnecessary Authorized Nuclear Inservice Inspector (ANII) involvement. As a result, the relief request remains valid and unaffected by the proposed change.

Surry engineering performs IWE/IWL ISI inspection activities in support of the required Type A (ILRT) test. There will be no change to the schedule for these inspections due to the extension of the Type A test interval. The activities performed that assure continued containment integrity include:

- During 1998 and 2000 refueling outages, Surry performed an IWE General Visual examination of the Containment Metal Liner (IWE - MC component). All accessible areas were examined. Although localized rust and surface anomalies were detected, no repairs were required to meet code requirements.

- Engineering inspected and evaluated the condition of the containment dome liner and determined that the coating continued to be acceptable for service. Inspections were completed on the containment dome liner above the polar crane line in November 1998 and followed up again in the spring of 2000. These inspections concluded that any observed peeling of the paint or rust was superficial in nature and no visible damage exists. No degradation was observed between inspections.

- Recent IWL containment ISI Program inspections of the Surry Units 1 and 2 containment structure identified embedded material in the containment dome areas. In the Unit 1 containment dome the material appears to be wood, approximately 2" x 12", with side grain exposed. The findings represent a direct inspection of approximately one-third of each containment dome and a remote inspection of the remainder of the containment dome.

  The embedded material, as described above, was inadvertently left in the containment structure during original plant construction. The slight depression in the location of the wood and below the adjacent concrete indicates that the wood was likely concealed below a thin layer of cement paste immediately following removal of the concrete form-work. Over time this thin layer of concrete has spalled off, leaving the wood exposed.

  Engineering performed an assessment of the significance of the embedded material. The assessment concluded that the containment structures remain fully capable of meeting the functional design requirements as described in Technical Specification 5.2-1 and UFSAR Section 15.5. This assessment assumed that the piece of wood extends from the concrete surface through the concrete placement and also assumed that similar embedded pieces of wood could exist in the two-thirds of the containment dome not directly inspected to date. An engineering evaluation of the inspection findings was performed and concluded that:

  - the leak-tight integrity of the liner has not been jeopardized,
  - any degradation of the underlying reinforcing steel as a result of the embedded wood is insignificant to the structure,
  - the loss of concrete displaced by the wood will have an insignificant effect upon the structure,
  - no significant loss of radiological shielding or missile protection has occurred.

  Based on the current inspections and the associated engineering evaluations to date, Dominion has not classified any areas as Examination Category E-C (accelerated degradation).

Evaluation of Significant Hazards Consideration

The proposed revision to Technical Specifications permits a one-time extension to the current interval for Type A testing. The current test interval of ten years, which is based on the standard of good past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Surry Unit 1. In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

1. Does the proposed license amendment involve a significant increase in the

probability or consequences of an accident previously evaluated?

The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence nor, is an activity or modification by itself that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG-1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and C tests. It concludes that reducing the Type A (ILRT) testing frequency to once per twenty years leads to an imperceptible increase in risk.

Surry provides a high degree of assurance through indirect testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptance criteria, indicating a very leak-tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak-tightness. Also, maintaining the containment subatmospheric during operations provides constant monitoring of the leaktightness of the containment structure. Separately, Type B and C testing, required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Surry Type A test interval will not represent a significant increase in the consequences of an accident.

2. Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed revision to Technical Specifications adds a one-time extension to the current interval for Type A testing for Surry Unit 1. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   The proposed revision to Surry Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Surry Unit 1. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in

Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal affect on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing. In addition, the risk impact on the total integrated (fifteen year total) Surry Unit 1 plant risk above baseline, for those accident sequences influenced by Type A testing, is only 0.004%. Furthermore, for Surry, maintaining the containment subatmospheric during plant operations further reduces the risk of any containment leakage path going undetected.

Implementation of the Proposed Change

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section IV of this evaluation, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupation radiation exposure.

The proposed change does not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Dominion concludes that the proposed changes meet the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.22 relative to requiring a specific environmental assessment by the Commission.

Conclusion

The proposed one-time change will not alter assumptions relative to the mitigation of an accident or transient event and will not adversely affect normal plant operation and testing. The proposed change is consistent with the current safety analysis assumptions and with the Technical Specifications. As such, no question of safety exists.

The Station Nuclear Safety and Operating Committee (SNSOC) and the Management Safety Review Committee (MSRC) have reviewed this proposed change to the TS and have concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

References

1. NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995.

2. NRC letter to Surry Issuing Technical Specification Amendment 208, dated April 18,1996 to implement the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary reactor containment leakage testing.

3. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

4. American National Standard ANSVANS - 56.8 - 1994, "Containment System Leakage Testing Requirements."

5. NUREG-1493, "Performance-Based Containment Leak-Test Program," Final Report, September 1995.

Attachment 2

Risk Assessment

Surry Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

VIRGINIA POWER		CALC ADDENDUM COVER SHEET
Calc Number: SM-1321	**Rev.** 0	**Add:** A
Station(s): Surry Power Station	**Unit 1 & 2**	**Sheet** 1 of 21

Addendum Title (Subject) Risk Impact Assessment of Extending Containment Type A Test Interval at Surry Power Station

Changes Calc Status? [ ] Yes [X] No New Status: ______

Reference Numbers: IR No.: ______ Job No. ______

Initiating Document: (DCP, IEER, REA, etc.): ______

Attachments? [X] Yes [ ] No Labeled A through B

Originator: [X] Virginia Power Discipline: Nuclear Analysis and Fuel

[ ] A/E Firm Name: ______

EDS Mark Number References: (Not listed in the Calc or previous addenda)

Station	Unit	System	Prefix	Sequence	Comp. Code	Suffix
[___]	[__]	[____]	[____]	[__________]	[________]	[__________]
[___]	[__]	[____]	[____]	[__________]	[________]	[__________]

[ ] Additional Mark Numbers? (Check if "yes"). See ______.

Objective:

Provide a risk impact assessment on extending the plant's integrated leak rate test (ILRT) interval from ten to fifteen years. The risk assessment is performed in accordance with the guidelines set forth in NEI 94-01 [1], the methodology used in EPRI TR-104285 [2] and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, RG 1.174 [3]. The purpose of this Addendum is to re-paginate pages that were in error in the original calculation, otherwise everything else is the same.

Conclusions:

The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years (5 year increase) increases the risk of those associated specific accident sequences by 0.001%. However, the risk impact on the total integrated (15 year increase) plant risk above baseline, for those accident sequences influenced by Type A testing is only 0.004%. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 4.37×10^{-8}/yr. Therefore, the risk impact when compared to other severe accident risks is negligible.

	Signature	Date
Prepared By (Print Name) Myron G. Matras	Myron G. Matras	9-20-01
Reviewed By (Print Name) Dave M. Bucheit	David M. Bucheit	9-20-01
Other (If Applicable Print Name)		

Table of Contents

1.0 Objective

In October 26, 1995, the NRC revised 10 CFR 50, Appendix J. The revision to Appendix J allowed individual plants to select containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements". The Surry Unit 1 and 2 Nuclear Power Plant selected the requirements under Option B as its testing program [4].

The surveillance testing requirements as proposed in NEI 94-01 [1] for Type A testing is at least once per 10 years based on an acceptable performance history (defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than $1L_a$).

Surry Unit 1 current ten-year Type A test is due to be performed during refueling outage (Cycle 19), scheduled for April 13, 2003. Unit 1 test was originally scheduled for fall of 2001 but will use the 15 month allowed one time extension. Surry Unit 2 Type A test is scheduled for year 2010.

This calculation will provide a risk impact assessment on extending the plant's integrated leak rate test (ILRT) interval from ten to fifteen years. The risk assessment will be performed in accordance with the guidelines set forth in NEI 94-01 [1], the methodology used in EPRI TR-104285 [2] and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, RG 1.174 [3]

In addition, the results and findings from the Surry Individual Plant Examination (IPE) [5] and revised model [10,17] are used for this risk assessment calculation.

2.0 Assumptions, Design Inputs and Key Parameter Uncertainties
2.1 Assumptions

The Surry leakage rate (L_a) acceptance criteria is defined as:
L_a = 0.1 percent by weight of containment air per 24 hours at calculated peak pressure (P_a).

1. Containment leak rates greater than $1L_a$, but less than 35 L_a, indicate an impaired containment. Break openings of greater than 0.1-inch and less than 0.7-inch in diameter are considered as small leak rate releases. Break openings of greater than 0.7-inch diameter are considered as large leak rate releases.

2. Containment leak rates greater than 35 L_a, indicate a containment breach. This leak rate is considered 'large'.

3. Containment leak rates less than $1L_a$ indicate an intact containment. This leak rate is considered as 'negligible'.

4. The maximum containment leakage for Class 1 sequences is $1L_a$.

5. The maximum containment leakage for Class 2 sequences is $35L_a$.

6. The maximum containment leakage for Class 3a sequences is $10L_a$.

7. The maximum containment leakage for Class 3b sequences is $35L_a$.

8. The maximum containment leakage for Class 6 sequences is $35L_a$.

9. Because Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.

10. Containment leakage due to Classes 4 and 5 are considered negligible based on the previously approved methodology [12].

11. The containment releases are not impacted with time.

12. The containment releases for Classes 2, 6, 7 and 8 are not impacted by the ILRT Type A Test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A.

2.2 Design Inputs

This calculation will use Surry 50 mile population data for calculating the population dose, which was also used for license extension SAMA analysis as discussed in Reference 6. The source term category (STC) release fractions and corresponding frequencies were taken from the Surry IPE report and revised data in Reference 17. Source term category is defined here as a grouping of like releases (CET endpoints) such that the offsite consequences are expected to be similar. There are enough STCs to cover the spectrum of releases.

2.3 Key Parameter Uncertainties

No parameter uncertainties were included in this calculation since only best estimate results are sought. The results of this calculation were not tested against any acceptance criteria because these accidents are beyond those used for licensing purposes as discussed in the UFSAR.

3.0 Method of Analysis

A simplified bounding analysis approach for evaluating the change in risk associated with increasing the interval from 10-years-to 15-years for Type A test was used. Type A test measures the containment air mass and calculates the leakage from the change in mass over time. This approach is similar to that presented in EPRI TR-104285 [2] and NUREG-1493 [7]. Namely, the analysis performed examined SPS IPE [5] plant specific accident sequences in which the containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or steam generator manway leakage. (EPRI TR-104285 Class 3 sequences). Type B test measures component leakage across pressure retaining boundaries (e.g. gaskets, expansion bellows and air locks. Type C test measures component leakage rates across containment isolation valves.
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left 'opened' following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences).
- Accident sequences involving containment failure induced by severe accident phenomena (EPRI TR-104285 Class 7 sequences), containment bypassed (EPRI TR-104285 Class 8 sequences) and large containment isolation failures (EPRI TR-104285 Class 2 sequences). Small containment isolation 'failure-to-seal' events (EPRI TR-104285 Class 4 and 5 sequences) were not accounted for in this evaluation. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test interval.

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 1. Map the Level 3 release categories into 8 release classes defined by the EPRI Report [2]. See Table A-1 of Attachment A.

Step 2 - Develop baseline plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285 [2].

Step 3 - Evaluate risk impact of extending Type A test interval from 10-to-15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [3]

Step 1 - Quantify the baseline risk in terms of frequency per reactor year.

This step involves the review of the Surry IPE [5] containment event tree (CET). The CET characterizes the response of the containment to important severe accident sequences. The CET used in this evaluation is based on important phenomena and systems-related events identified in NUREG-1335 [8] and NSAC-159, Volume 2 [9] and on plant features that influence the phenomena.

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing or containment failure induced by severe accident phenomena. As a result, the CET containment isolation model was reviewed for applicable isolation failures and their impact on the overall plant risk.

The containment isolation model found in References 10 & 11 examined the five issues associated with containment isolation in NUREG-1335 [8]. These issues are:

(1) The identity of pathways that could significantly contribute to containment isolation failure.
(2) The signals required to automatically isolate the containment penetration.

(3) The potential generating signals for all initiating events.
(4) The examination of testing and maintenance procedures.
(5) The quantification of each containment isolation mode.

The containment isolation model in References 10 & 11 screened out lines less than 5.5 inches in diameter which was the minimum cutoff for the LERF definition. This evaluation considers lines between 0.1 inches and 5.5 inches as potential candidates for significant containment leakage.

The Level 3 release categories were mapped into 8 release classes (See Table A-1 in Attachment A) as defined in the EPRI Report [2]. These EPRI containment failure classifications are listed below.

EPRI Containment Failure Classifications

Class 1 Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant. The allowable leakage rates (L_a), are typically 0.1 weight percent of containment volume per day for PWRs and 0.5 weight percent per day for BWRs (all measured at P_a, calculated peak containment pressure related to the design basis accident). Changes to leak rate testing frequencies do not affect this classification.

Class 2 Containment isolation failures (as reported in the IPEs) include those accidents in which the pre- existing leakage is due to failure to isolate the containment. These include those that are dependent on the core damage accident in progress (e. g., initiated by common cause failure or support system failure of power) and random failures to close a containment path. Changes in Appendix J testing requirements do not impact these accidents.

Class 3 Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i. e., provide a leak-tight containment) is not dependent on the sequence in progress. This accident class is applicable to sequences involving ILRTs (Type A tests) and potential failures not detectable by LLRTs.

Class 4 Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B- tested components that have isolated but exhibit excessive leakage.

Class 5 Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.

Class 6 Containment isolation failures include those leak paths not identified by the LLRTs. The type of penetration failures considered under this class includes those covered in the plant test and maintenance requirement or verified by in service inspection and testing (ISVIST) program. This failure to isolate is not typically identified in LLRT. Changes in Appendix J LLRT test intervals do not impact this class of accidents.

Class 7 Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.

Class 8 Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in class 8. Changes in Appendix J testing requirements do not typically impact these accidents, particularly for PWRs.

The frequencies for the above eight classes are calculated below. The Class 3–6 frequencies are calculated first since these values are needed to determine the Class 1 frequency.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (i.e. containment liner) exists. The containment leakage for these sequences can be either small (1L_a to 35 L_a) or large (>35 L_a).

To calculate the probability that a liner leak will be large (Event CLASS-3B), use was made of the data presented in NUREG-1493 [7]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release (refer to the write-up in Step 4), no large releases have occurred based on the 144 ILRTs reported in NUREG-1493 [7].

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the X^2 distribution. In statistical theory, the X^2 distribution can be used for statistical testing, goodness-of-fit tests, and evaluating s-confidence [13]. The X^2 distribution is really a family of distributions, which range in shape from that of the exponential to that of the normal distribution. Each distribution is identified by the degrees of freedom, **v**. For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the X^2 distribution can be calculated as X^2_{95} (**v** = 2n+2)/2N, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks (n = 0) in 144 events (N = 144) and $X^2_{95}(2) = 5.99$, the 95th percentile estimate of the probability of a large leak is calculated as 5.99/(2*144) = 0.021.

To calculate the probability that a liner leak will be small (Event CLASS-3A), use was made of the data presented in NUREG-1493 [7]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of 1.0L_a. However, of these 23 'failures' only 4 were found by an ILRT, the others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for "small releases" are 4-of-144. Similar to the event CLASS-3B probability, the estimated failure probability for small release is found by using the X^2 distribution. The X^2 distribution is calculated by n=4 (number of small leaks) and N=144 (number of events) which yields a X^2 (10) =18.3070. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as 18.3070/(2*144) = 0.064.

The respective frequencies per year are determined as follows:

$$\text{CLASS-3A-FREQUENCY} = \text{PROB}_{\text{class-3a}} * \text{CDF}$$
$$\text{CLASS-3B-FREQUENCY} = \text{PROB}_{\text{class-3b}} * \text{CDF}$$

Where:

$\text{PROB}_{\text{class-3a}}$ = probability of small pre-existing containment liner leakage

= 0.064

$PROB_{class\text{-}3b}$ = probability of large pre-existing containment liner leakage
= 0.021

CLASS-3A-FREQUENCY = 0.064 * 3.78 x 10^{-5} /year = 2.41 x 10^{-6} /year

CLASS-3B-FREQUENCY = 0.021 * 3.78 x 10^{-5} /year = 7.94 x 10^{-7} /year

For this analysis the associated maximum containment leakage for class 3A is $10L_a$ and for class 3B is $35L_a$

Class 4 Sequences. This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type B test components occurs. By definition these failures are dependent on Type B testing, and the probability will not be impacted by type A testing. Because these failures are detected by Type B tests, this group is not evaluated any further, consistent with approved methodology.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type C test components occurs. By definition these failures are dependent on Type C testing, and the probability will not be impacted by type A testing. Because these failures are detected by Type C tests, this group is not evaluated any further, consistent with approved methodology.

Class 6 Sequences. This group is similar to Class 2 and addresses additional failure modes not typically modeled in PRAs due to the low probability of occurrence. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. A conservative screening value of 1.0E-03 will be used to evaluate this class.

The low failure probabilities are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the purpose of this calculation, and the fact that this failure class is not impacted by Type A testing, no further evaluation is needed. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology (i.e.-PROBclass6 > 0), a conservative screening value of I.0E-03 will be used to evaluate this class.

FREQclass6 = (Screening Value) x CDF

Screening Value = 1.0 x 10^{-3} [Assumed Conservative Value]

CLASS-6-FREQUENCY = 1.0 x 10^{-3} * 3.78 x 10^{-5} /year = 3.78 x 10^{-8} /year

For this analysis the associated maximum containment leakage for this group is $35L_a$

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact. The frequency per year for these sequences is 1.73×10^{-5} /year (Attachment A, Table A-1). For this analysis the associated maximum containment leakage for this group is $1L_a$. The Surry IPE did not model Class 3 or Class 6 type failures, therefore this needs to be accounted for in the Class 1 accident class. Using Reference 16 methodology, the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a, Class 3b and Class 6 in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$$\text{CLASS-1-FREQ} = \text{FREQ}_{\text{Class-1}} - (\text{FREQ}_{\text{Class3a}} + \text{FREQ}_{\text{Class3b}} + \text{FREQ}_{\text{Class6}})$$

$$\text{CLASS-1-FREQ} = 1.73 \times 10^{-5} - (2.41 \times 10^{-6} + 7.94 \times 10^{-7} + 3.78 \times 10^{-8})$$

CLASS-1-FREQ =1.41 x 10^{-5} /year

Class 2 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isloate the containment occurs. The frequency for Class 2 is the sum of those release categories identified in Table A-1 as Class 2.

CLASS-2-FREQUENCY = 4.91 x 10^{-6} /year [Table A-1]

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena (Early and Late Failures). The frequency of Class 7 is the sum of those release categories identified in Table A-1 as Class 7.

CLASS-7-FREQUENCY = 11.65 x 10^{-6} / year

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. The frequency of Class 8 is the sum of those release categories identified in Table A-1 as Class 8.

CLASS-8-FREQUENCY = 3.94 x 10^{-6} / year

Note: for this class the maximum release is not based on normal containment leakage, because the releases are released directly to the environment. Therefore, the containment structure will not impact the release magnitude.

Table 1
Mean Containment Frequencies Measures - Given Accident Class

Class	Description	Frequency (per Rx-year)
1	No Containment Failure	1.41×10^{-5}
2	Large Containment Isolation Failures (Failure-to-close)	4.91×10^{-6}
3a	Small Isolation Failures (Type A test)	2.41×10^{-6}
3b	Large Isolation Failures (Type A test)	7.94×10^{-7}
4	Small isolation failure - failure-to-seal (Type B test)	Not Analyzed
5	Small isolation failure - failure-to-seal (Type C test)	Not Analyzed
6	Containment Isolation Failures (dependent failures, personnel errors)	3.78×10^{-8}
7	Severe Accident Phenomena Induced Failure (Early and late Failures)	11.65×10^{-6}
8	Containment Bypassed (SGTR & V-Sequence)	3.94×10^{-6}
Core Damage	Core Damage All CET End states	3.78×10^{-5}

Step 2 – Develop baseline plant specific person-rem dose (population dose) per reactor year.

Plant-specific MAAP/MACCS2 analysis was performed to evaluate the person-rem dose to the population, within a 50-mile radius from the Surry plant. Since a Class 1 dose was not available, the Source Term Category 2 (Class 7) release was used for Class 1 accident release, see Table A-1 in Attachment A. The Source Term Category 2 accident sequence MAAP run has characteristics that are representative of an EPRI Class 1 containment leakage. The Source Term Category 2 (Class 7) accident sequence is conservative to use in place of the Category 1 (Class 1) sequence which assumes a much smaller containment leak rate. The dose for Class 2 accidents is the sum of the Class 2 dose values from Table A-1.

Using the total population dose for Class 1 accident as the starting reference point, the Class 3 through 6 accidents are calculated below. The population dose is converted to the corresponding Class value using the appropriate dose multiplier as was used in Reference 12 to predict the person-rem dose for accident classes 1 to 6 as follows. The dose for Class 7 accidents is the sum of all Class 7 accidents having CDF greater than zero from Table A-1.

Class 1 = 5.98×10^2 person-rem
Class 2 = 2.22×10^6 person-rem
Class 3a = $5.98 \times 10^2 * 10\ L_a = 5.98 \times 10^3$ person-rem
Class 3b = $5.98 \times 10^2 * 35\ L_a = 2.09 \times 10^4$ person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = $5.98 \times 10^2 * 35L_a = 2.09 \times 10^4$ person-rem
Class 7 = 2.15×10^6 person-rem

Class 8 sequences include containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are expected to be released directly to the environment. The sum of Class 8 dose from Table A-1 represent the sum of the dose for the Event-V and SGTR sequences. The total population dose for Class 8 bypass sequence is 1.46E+07 person-rem.

The above values are summarized in Table 2 below.

Table 2
Person-Rem Measures - Given Accident Class

Class	Description	Person-Rem (50-Miles)
1	*No Containment Failure	5.98×10^{2}
2	Large Containment Isolation Failures (Failure-to-close)	2.22×10^{6}
3a	Small Isolation Failures (Type A test)	5.98×10^{3}
3b	Large Isolation Failures (Type A test)	2.09×10^{4}
4	Small isolation failure - failure-to-seal (Type B test)	N/A
5	Small isolation failure - failure-to-seal (Type C test)	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	2.09×10^{4}
7	Failure Induced by Phenomena (Early and Late Failures)	2.15×10^{6}
8	Containment Bypassed (SGTR & V-Sequence)	1.46×10^{7}

*Note: Release Category 2 dose from Table A-1 of Attachment A was used for Class 1 accident.

The above dose results when combined with the frequency results presented in Table 1 yields the SPS baseline mean consequence measures for each accident class. These results are presented in Table 3 below.

Table 3
Baseline Mean Person-Rem Measures - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	1.41×10^{-5}	5.98×10^{2}	8.432×10^{-3}
2	Large Isolation Failures (Failure-to-close)	4.91×10^{-6}	2.22×10^{6}	10.900
3a	Small Isolation Failures (Type A test)	2.41×10^{-6}	5.98×10^{3}	1.441×10^{-2}
3b	Large Isolation Failures (Type A test)	7.94×10^{-7}	2.09×10^{4}	1.659×10^{-2}
4	Small isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	N/A
5	Small isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	3.78×10^{-8}	2.09×10^{4}	7.900×10^{-4}
7	Failure Induced by Phenomena (Early and Late Failures)	11.65×10^{-6}	2.15×10^{6}	25.048
8	Containment Bypassed (SGTR & V-Sequence)	3.94×10^{-6}	1.46×10^{7}	57.524
CDF	All CET End States	3.78×10^{-5}	N/A	93.512

Based on the above values, using the same methodology as Reference 16, the baseline percent risk contribution due to Type A testing is as follows:

$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS3b_{BASE}) / Total_{BASE}] \times 100$

Where:

$CLASS3a_{BASE}$ = class 3a person-rem/year = 1.441 x 10^{-2} person-rem/year [Table 3]

$CLASS3b_{BASE}$ = class 3b person-rem/year = 1.659 x 10^{-2} person-rem/year [Table 3]

$Total_{BASE}$ = total person-rem/year for baseline interval = 93.512 person-rem/year [Table 3]

%RiskBASE =[(1.441 x 10^{-2} + 1.659 x 10^{-2}) / 93.512] x 100 %
RiskBASE = 0.033%

Therefore, the baseline percent risk contribution, due to Type A testing is 0.033%.

Step 3 - Evaluate risk impact of extending Type A test interval from 10-to-15 years.

According to NUREG-1493 [7], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. (The average time for undetection is calculated by multiplying the test interval by ½ and multiplying by 12 to convert from years" to "months"). If the test interval is extended to 1 in 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (1/2* 15 * 12). Since ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% increase in the overall probability of leakage. This value is determined by multiplying 3% and the ratio of the average time for undetection for the increased ILRT test interval (60 months) to the baseline average time for undetection of 18 months. For a 15-yr-test interval, the result is a 15% increase in the overall probability of leakage (i.e., 3 * 90/18).

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class 1 and Class 3 sequences. In addition the increased probability of not detecting excessive leakage has no impact on the frequency of occurrence for Class 1 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large liner opening remains the same, even though the probability of not detecting the liner opening increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 1.1. (Recall that for a 10-year interval there is a 10% increase on the overall probability of leakage).

The frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a and Class 3b in order to preserve the total CDF. The revised Class 1 frequency is therefore:

CLASS-1-FREQ = $FREQ_{Class-1}$ - ($FREQ_{Class3a}$ + $FREQ_{Class3b}$)*0.1

CLASS-1-FREQ = 1.41 x 10^{-5} - (2.41 x 10^{-6} + 7.94 x 10^{-7}) *0.1

CLASS-1-FREQ =1.38 x 10^{-5} /year

The results of this calculation are presented in Table 4 below.

Table 4
Mean Consequence Measures for 10-Year Test Interval - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	1.38×10^{-5}	5.98×10^{2}	8.252×10^{-3}
2	Large Isolation Failures (Failure-to-close)	4.91×10^{-6}	2.22×10^{6}	10.900
3a	Small Isolation Failures (Type A test)	2.65×10^{-6}	5.98×10^{3}	1.585×10^{-2}
3b	Large Isolation Failures (Type A test)	8.73×10^{-7}	2.09×10^{4}	1.825×10^{-2}
4	Small isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	N/A
5	Small isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	3.78×10^{-8}	2.09×10^{4}	7.900×10^{-4}
7	Failure Induced by Phenomena (Early and Late Failures)	11.65×10^{-6}	2.15×10^{6}	25.048
8	Bypass (SGTR)	3.94×10^{-6}	1.46×10^{7}	57.524
CDF	All CET End States	3.78×10^{-5}	N/A	93.515

Based on the above values, the Type A 10-year test frequency percent risk contribution ($\%Risk_{10}$) for Class 3 is as follows:

$$\%Risk_{10} = [(CLASS3a_{10} + CLASS3b_{10}) / Total_{10}] \times 100$$

Where:

$CLASS3a_{10}$ = class 3a person-rem/year = 1.585×10^{-2} person-rem/year [Table 4]

$CLASS3b_{10}$ = class 3b person-rem/year = 1.825×10^{-2} person-rem/year [Table 4]

$Total_{10}$ = total person-rem year for 10-year interval = 93.515 person-rem/year [Table 4]

$$\%Risk_{10} = [(1.585 \times 10^{-2} + 1.825 \times 10^{-2}) / 93.515] \times 100$$
$$\%Risk_{10} = 0.036\%$$

Therefore, the total 10-year test frequency ILRT interval percent risk contribution due to Type A testing is 0.036%.

<u>The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is as follows:</u>

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE} \times 100.0$$

Where:

$Total_{BASE}$ =total person-rem/year for baseline interval = 93.512person-rem/year [Table 3]

$Total_{10}$ = total person-rem/year for 10-year interval = 93.515 person-rem/year [Table 4]

$$\Delta\%Risk_{10} = [(93.515 - 93.512) / 93.512] \times 100.0$$
$$\Delta\%Risk_{10} = 0.003\%$$

Therefore, the increase in risk contribution because of relaxed ten-year ILRT test frequency from three-in-ten-years to 1-in-ten-years is 0.003%.

Risk Impact due to 15-year Test Interval

The risk contribution for a 15-year interval is similar to the 10-year interval. The difference is in the increase in probability of leakage value. For this case the value is 15 percent or 1.15. (Recall that for a 10-year interval there is a 10% increase on the overall probability of leakage). In addition, the containment leakage used for the 10-year test interval for both Class 1 and Class 3 are used in the 15-year interval evaluation.

The frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a and Class 3b in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$$\text{CLASS-1-FREQ} = FREQ_{Class\text{-}1} - (FREQ_{Class3a} + FREQ_{Class3b})*0.15$$

$$\text{CLASS-1-FREQ} = 1.41 \times 10^{-5} - (2.41 \times 10^{-6} + 7.94 \times 10^{-7}) *0.15$$

CLASS-1-FREQ =1.36 x 10^{-5} /year

The results of this calculation are presented in Table 5 below.

Table 5
Mean Consequence Measures for 15-Year Test Interval - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	1.36×10^{-5}	5.98×10^{2}	8.133×10^{-3}
2	Large Isolation Failures (Failure-to-close)	4.91×10^{-6}	2.22×10^{6}	10.900
3a	Small Isolation Failures (Type A test)	2.77×10^{-6}	5.98×10^{3}	1.656×10^{-2}
3b	Large Isolation Failures (Type A test)	9.13×10^{-7}	2.09×10^{4}	1.908×10^{-2}
4	Small isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	N/A
5	Small isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	3.78×10^{-8}	2.09×10^{4}	7.900×10^{-4}
7	Failure Induced by Phenomena (Early and Late Failures)	11.65×10^{-6}	2.15×10^{6}	25.048
8	Bypass (SGTR)	3.94×10^{-6}	1.46×10^{7}	57.524
CDF	All CET End States	3.78×10^{-5}	N/A	93.516

Based on the above values, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class 3 is as follows:

$$\%Risk_{15} = [(CLASS3a_{15} + CLASS3b_{15}) / Total_{15}] \times 100$$

Where:
$CLASS3a_{15}$ = class 3a person-rem/year = 1.656×10^{-2} person-rem/year [Table 5]

$CLASS3b_{15}$ = class 3b person-rem/year = 1.908×10^{-2} person-rem/year [Table 5]

$Total_{15}$ = total person-rem year for 15-year interval = 93.516 person-rem/year [Table 5]

$\%Risk_{15} = [(1.656 \times 10^{-2} + 1.908 \times 10^{-2}) / 93.515] \times 100$ %
$\%Risk_{15} = 0.038\%$
Therefore, the total Type A 15-year ILRT interval risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 0.038%.

<u>The percent increase on the total integrated plant risk due to a five-year increase over the 10 year ILRT is computed as follows.</u>

$\%TOTAL_{10\text{-}15} = [(TOTAL_{15} - TOTAL_{10}) / TOTAL_{10}] \times 100$

Where:
$TOTAL_{10}$ = total person-rem/year for 10-year interval = 93.515 person-rem/year [Table 4]
$TOTAL_{15}$ = total person-rem/year for 15-year interval = 93.516 person-rem/year [Table 5]

$\%TOTAL_{10\text{-}15} = [(93.516 - 93.515)/ 93.515] \times 100$
$\%TOTAL_{10\text{-}15} = 0.001\%$

Therefore, the risk impact on the total integrated plant risk for these accident sequences influenced by Type A testing is only 0.001%.

<u>The percent risk increase ($\Delta\%Risk_{15}$) due to a fifteen-year ILRT over the baseline case is as follows:</u>

$\Delta\%Risk_{15} = [(TOTAL_{15} - Total_{BASE}) / Total_{BASE} \times 100$

Where:
$Total_{BASE}$ = total person-rem/year for baseline interval = 93.512 person-rem/year [Table 3]
$Total_{15}$ = total person-rem/year for 15-year interval = 93.516 person-rem/year [Table 5]

$\Delta\%Risk_{15} = [(93.516 – 93.512) / 93.512] \times 100.0$
$\Delta\%Risk_{15} = 0.004\%$

Therefore, the total increase in risk contribution associated with relaxing the ILRT test frequency from three in ten years to once-per-fifteen years is 0.004%.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF)

The one time extension of increasing the Type A test interval involves establishing the success criteria for a large release. This criteria is based on two prime issues:

1) The containment leak rate versus breach size, and

2) The impact on risk versus leak rate.

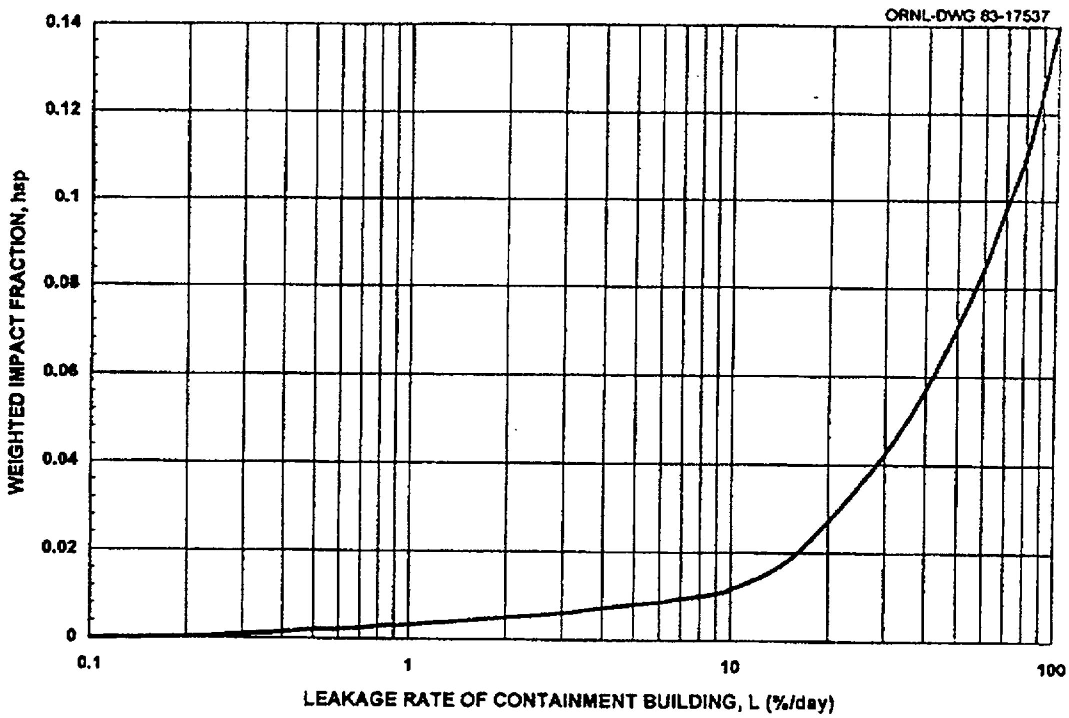
The containment leak size for the corresponding leak rate was calculated using the same methodology as in Reference 10. The effect of containment leak size on the containment leak rate is shown in Table 6. In addition, Oak Ridge National Laboratory (ORNL) [14] completed a study evaluating the impact of leak rates on public risk using information from WASH-1400 [15] as the basis for its risk sensitivity calculations (see Figure 1).

Based upon the information in Table 6 and ORNL, it is judged that small leaks resulting from a severe accident (that are deemed not to dominate public risk) can be defined as those that change risk by less than 5%. This definition would include leaks of less than 35%/day. Based on the Table 6 data, a 35%/day containment leak rate equates to a diameter leak of slightly smaller than 0.7 inches. It is to be noted that for Surry a containment diameter of 0.7 inches was calculated as opposed to 2.0 inches for Indian Point 3. This difference in containment leak diameter is due to the difference in containment size between Surry and Indian Point 3. Therefore, this study defines small leakage as containment leakage resulting from an opening of 0.36 in^2 or less, large leakage as greater than 35%/day and negligible leakage as 0. 1% /day to 2%/day.

Table 6
Evaluated Impact of Containment Leak Size on Containment Leak Rate

Containment Leak Size		**Approximate Containment Leak Rate at Design Pressure**
Diameter (inches)	**Area (in^2)**	**(wt%/day)**
0.036	0.001	0.1
0.115	0.010	1.0
0.364	0.104	10.0
0.681	0.363	35.0
1.152	1.043	100.0
5.647	25.05	2400

Figure I
Fractional Impact on Risk Associated with Containment Leak Rates [14]



The risk impact associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a large release due to failure to detect a pre-existing leak during the relaxation period. For this evaluation only Class 3 sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because for these sequences the containment remains intact. Therefore, the containment leak rate is expected to be small (less than $2L_a$. A larger leak rate would imply an impaired containment, such as classes 2, 3, 6 and 7.

Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event. At the same time, sequences in the Surry IPE [5], which result in large releases (e.g., large isolation valve failures), are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, the frequency of Class 3B sequences (Table 4) is used as the LERF for Surry. This frequency, based on a ten-year test interval, is 8.73×10^{-7}/yr.

Reg. Guide 1.174 [3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 [3] defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

As described in Step 3, extending the ILRT interval from once-per-10 years to once-per-15 years will increase the average time that a leak detectable only by an ILRT goes undetected from 60 to 90 months (0.5*15*12). Since ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 15-yr ILRT interval is a 15% increase in the overall probability of leakage (3 * 90/18) versus 10% for a 10-yr ILRT interval. Thus, increasing the ILRT test interval from 10 years to 15 years results in a 5% increase in the overall probability of leakage. Multiplying the above LERF frequency (8.73×10^{-7}/yr) by the increase in overall probability of leakage (0.05) gives an increase in LERF of 4.37×10^{-8}/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1 E-7/yr, increasing the ILRT interval to 15 years is non-risk significant.

It should be noted that if the risk increase is measured from the original 3-in-10-year interval, the increase in LERF is 7.94×10^{-7}/yr from Class 3B sequences (Table 3) multiplied by the 12% incremental increase in overall probability for a fifteen-year test interval (i.e., 15% - 3%) is 9.53×10^{-8}/yr, which is slightly below the 1.0E-7/yr screening criterion in Reg. Guide 1.174).

4.0 DATA Used

The revised frequency data was taken from Reference 10, which is shown on Table A-1 in Attachment A. The person-Rem dose data corresponding to the source term category was taken from the Level 3 SAMA analysis documented in Reference 6. Some data and some assumptions were taken from the Indian Point 3 evaluation in Reference 12. Some methodology and data was used from the Crystal River evaluation documented in Reference 16 as noted.

5.0 Computer Codes And Computer Used

This calculation did not generate output that required the use of computer codes. Some information used within this calculation was generated by computer codes. These computer codes include the MAAP, MACCS2, NUPRA and NUCAP codes. The MAAP code was used to generate source term release fractions and the MACCS2 code was used to generate person-rem dose. The NUPRA and NUCAP codes were used to generate the Plant Damage States, Source Term Categories and corresponding frequencies respectively. The MAAP code was run on the IBM mainframe. The MACCS2, NUPRA and NUCAP codes were run on a desktop personal computer.

6.0 Results

1. The baseline percent risk contribution of Type A testing, represented by Class 3 accident scenarios is 0.033%.

2. Type A 10-year ILRT interval percent risk contribution of Type A testing, represented by Class 3 accident scenarios is 0.036%.

3. Type A 15-year ILRT interval percent risk contribution of leakage, represented by Class 3 accident scenarios is 0.038%.

4. The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-ten-year interval to once-per-fifteen years (a 5 year increase) is 0.001%

5. The total integrated increase in risk contribution from extending the ILRT test frequency from the baseline case, 3 in ten years interval, to once-per-15 years (a 15 year increase) is 0.004%.

6. The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 4.37×10^{-8}/yr.

7. The risk increase in LERF from the original 3-in-10-year interval, to once-per-15 years is 9.53×10^{-8}/yr

8. Other salient results are summarized in Table 7.

Table 7
Summary of Risk Impact on Extending Type A ILRT Test Frequency

Class*	Risk Impact (BASE)**	Risk Impact (10-years)***	Risk Impact (15-years)****
3a and 3b	0.033% of integrated value based on $10L_a$ Class 3a and $35L_a$, for Class 3b	0.036% of integrated value based on $10L_a$ for Class 3a and $35L_a$ for Class 3b	0.038% of integrated value based on $10L_a$, for Class 3a and $35L_a$ for Class 3b
Total Integrated Risk	93.512 person-rem/year	93.515 person-rem/year	93.516 person-rem/year

* Only accident sequences impacted by a change in Type A test frequency are evaluated. These are sequences 3A and 3B
** SPS Revised IPE baseline values
***Type A ILRT test interval of 1-in-10-years
****Type A ILRT test interval of 1-in-15-years

7.0 Conclusions

Based on the above results, the following are conclusions regarding the assessment of the plant risk associated with extending the Type A ILRT test frequency from ten-years to fifteen years.

The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years (a 5 year increase) increases the risk of those associated specific accident sequences by 0.001%. However, the risk impact on the total integrated (15 year increase) plant risk above baseline, for those accident sequences influenced by Type A testing is only 0.004%. Therefore, the risk impact when compared to other severe accident risks is negligible.

Reg. Guide 1. 174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 4.37×10^{-8}/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant.

8.0 References

1) RF-Report, NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 26, 1995, Revision 0
2) RF-Report, EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals" August 1994
3) RF-Report, Regulatory Guide 1.174, " An Approach for Using Probabilistic Risk Assessment In Risk-informed Decisions On Plant-Specific Changes to the Licensing Basis" July 1998.
4) RF-Procedure, Engineering Periodic Test 2-NPT-CT-101, "Reactor Containment Building Integrated Leak Rate Test (Type A Containment Testing)", Revision 5.
5) RF-Report, "Probabilistic Risk Assessment for The Individual Plant Examination Final Report", Surry Units 1 and 2, August 1991
6) RF-Calc., SM-1241 Revision 0, "MACCS2 model for Surry Level 3 Application", 2-28-2000.
7) RF-Report, NUREG-1493, "Performance-Based Containment Leak-Test Program, July 1995.
8) RF-Report, United States Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.
9) RF-Report, Z. T. Mendoza, et al., "Generic Framework for Individual Plant Examination (IPE) Backend (Level 2) Analysis, Volume 1 - Main Report and Volume 3 - BWR Implementation Guidelines," prepared by SAIC International, Inc., Electrical Power Research Institute, NSAC-159, EPRI PR3114-29, 1991.
10) RF-Calc., SM-1237 Revision 0, "Surry and North Anna Containment Isolation Modeling",4-20-00.
11) RF-Calc., SM-1237 Revision 0, Addendum A "Surry and North Anna Containment Isolation Modeling", 4-24-01.
12) RF-Calc., IP3-CALC-VC-03357 Revision 0,"Risk Impact Assessment of Extending Containment Type A Test Interval", 1-4-01.
13) RF-Report, Patrick D. T. O'Connnor, "Practical Reliability Engineering," John Wiley & Sons, 2nd Edition, 1985.
14) RF-Report, Burns, T.J., "Impact of Containment Building Leakage on LWR Accident Risk", Oak Ridge National Laboratory, NUREG/CR-3539, April 1984.
15) RF-Report, United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.
16) RF-Calc., Florida Power Calculation, F-01-0001, Revision 2, "Evaluation of Risk Significance of ILRT Extension", 6-19-01.

17) RF-Calc., SM-1256 Revision 0, “Surry Severe Accident Mitigation Alternative (SAMA)”,9-05-00.

Attachment A
STC/Frequency Data
Surry Units 1 and 2

Table A-1
Surry Frequency and Dose Data

Release Category	Frequency* Per year	Person-Rem**	EPRI Class	Description
1	1.73E-05	+5.98E+02	1	No CF
2	0.00E+00	5.98E+02	7	Early CF
3	0.00E+00	++8.23E+05	7	Early CF
4	0.00E+00	++1.74E+06	7	Early CF
5	0.00E+00	8.23E+05	7	Early CF
6	0.00E+00	++1.74E+06	7	Early CF
7	0.00E+00	2.59E+06	7	Early CF
8	0.00E+00	1.74E+06	7	Early CF
9	2.50E-06	+2.50E+04	7	Late CF
10	1.62E-07	+8.23E+05	7	Late CF
11	1.38E-07	2.50E+04	7	Late CF
12	8.91E-08	++8.23E+05	7	Late CF
13	4.92E-08	2.89E+05	7	Late CF
14	5.22E-06	++7.10E+04	7	Late CF
15	3.26E-06	7.10E+04	7	Late CF
16	2.28E-07	++2.50E+04	7	Melthru
17	0.00E+00	++5.98E+02	2	No Cont. Iso
18	1.28E-07	4.71E+05	2	No Cont. Iso
19	0.00E+00	++1.74E+06	2	Alpha CF
20	4.78E-06	+++1.19E+04	2	Debris Cool IV
21	0.00E+00	-------	1	Debris Cool IV
22	1.37E-06	2.75E+06	8	V-Sequence
23	2.42E-07	6.81E+06	8	V-Sequence
24	2.33E-06	5.07E+06	8	SGTR
CDF Freq	3.78E-05			

*Frequency data taken from Reference 17.
**Person-Rem data taken from Reference 6.
+ Used same dose as STC 2 (MAAP run has characteristics that are representative of an EPRI Class 1 containment leakage).
++Recommended Alternate values were used consistent with the IPE and SAMA analysis
+++ Use IPE STC 20 instead of STC 21 based on review of MAAP runs.
Total Class 1 Frequency = 1.73E-05 yr^{-1}
Total Class 2 Frequency = 4.91E-06 yr^{-1}
Total Class 7 Frequency = 11.65E-06 yr^{-1}
Total Class 8 Frequency = 3.94E-06 yr^{-1}
Total Class 2 Dose = 2.22E+06 Person-Rem
Total Class 7 Dose = 2.15E+06 Person-Rem
Total Class 8 Dose = 1.46E+07 Person-Rem

Dominion

CALCULATION REVIEW CHECKLIST **ATTACHMENT _B_**

Calculation No.SM-1321	Rev.0	Addendum	Page 1 of _1_

NOTE: If "Yes" is not answered, an explanation shall be provided below.
Reference may be made to explanations contained in the calculation or addendum.

Questions:	Yes	N/A
1. Have the sources of design inputs been correctly selected and referenced in the calculation?	[X]	[ ]
2. Are the sources of design inputs up-to-date and retrievable/attached to the calculation?	[X]	[ ]
3. Where appropriate, have the other disciplines reviewed or provided the design inputs for which they are responsible?	[X]	[ ]
4. Have design inputs been confirmed by analysis,test, measurement, field walkdown, or other pertinent means as appropriate for the configuration analyzed?	[X]	[ ]
5. Are assumptions adequately described and bounded by the Station Design Basis?	[X]	[ ]
6. Have the bases for engineering judgments been adequately and clearly presented?	[X]	[ ]
7. Were appropriate calculation/analytic methods used and are outputs reasonable when compared to inputs?	[ ]	[X]
8. Are computations technically accurate?	[X]	[ ]
9.Has the calculation made appropriate allowances for instrument errors and calibration equipment errors? (Reference STD-EEN-0304)	[ ]	[X]
10. Have those computer codes used in the analysis been referenced in the calculation?	[X]	[ ]
11. Have all exceptions to station design basis criteria and regulatory requirements been identified and justified in accordance with ANSI N45.2.11-1974?	[X]	[ ]

Comments: (Attach additional pages if needed)
Reviewer comments have been discussed and incorporated into the body of this calculation.

Prepared By (Print Name) Myron G. Matras	Signature	Date 09/20/2001
Reviewed By (Print Name) Dave Bucheit	Signature	Date 09/20/2001

Attachment 3

Mark-up of Unit 1 Technical Specifications Change

Surry Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage testing.

Objective

To assure that leakage of the primary reactor containment and associated systems is held within allowable leakage rate limits; and to assure that periodic surveillance is performed to assure proper maintenance and leak repair during the service life of the containment.

Specification

A. Periodic and post-operational integrated leakage rate tests of the containment shall be performed in accordance with the requirements of 10 CFR 50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

B. Containment Leakage Rate Testing Requirements

1. The containment and containment penetrations leakage rate shall be demonstrated by performing leakage rate testing as required by 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995, (2) Leakage rate acceptance criteria are as follows:

   a. An overall integrated leakage rate of less than or equal to La, 0.1 percent by weight of containment air per 24 hours, at calculated peak pressure (Pa).

   b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C testing when pressurized to Pa.

   Prior to entering an operating condition where containment integrity is required the as-left Type A leakage rate shall not exceed 0.75 La and the combined leakage rate of all penetrations subject to Type B and C testing shall not exceed 0.6 La.

3. ~~2.~~ The provisions of Specification 4.0.2 are not applicable.

Basis

The leak tightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.

AS MODIFIED BY THE FOLLOWING EXCEPTION:

NEI 94-01-1995, SECTION 9.2.3: THE FIRST UNIT 1 TYPE A TEST PERFORMED AFTER THE APRIL 23, 1992 TYPE A TEST SHALL BE PERFORMED NO LATER THAN APRIL 22, 2007.

Attachment 4

Proposed Unit 1 Technical Specifications Changes

Surry Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage testing.

Objective

To assure that leakage of the primary reactor containment and associated systems is held within allowable leakage rate limits; and to assure that periodic surveillance is performed to assure proper maintenance and leak repair during the service life of the containment.

Specification

A. Periodic and post-operational integrated leakage rate tests of the containment shall be performed in accordance with the requirements of 10 CFR 50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors."

B. Containment Leakage Rate Testing Requirements

1. The containment and containment penetrations leakage rate shall be demonstrated by performing leakage rate testing as required by 10 CFR 50 Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September, 1995 as modified by the following exception:

   NEI 94-01-1995, Section 9.2.3: The first Unit 1 Type A test performed after the April 23, 1992 Type A test shall be performed no later than April 22, 2007.

2. Leakage rate acceptance criteria are as follows:

   a. An overall integrated leakage rate of less than or equal to La, 0.1 percent by weight of containment air per 24 hours, at calculated peak pressure (Pa).

   b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C testing when pressurized to Pa.

   Prior to entering an operating condition where containment integrity is required the as-left Type A leakage rate shall not exceed 0.75 La and the combined leakage rate of all penetrations subject to Type B and C testing shall not exceed 0.6 La.

3. The provisions of Specification 4.0.2 are not applicable.

Basis

The leak tightness testing of all liner welds was performed during construction by welding a structural steel test channel over each weld seam and performing soap bubble and halogen leak tests.